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Indiana Michigan Power
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September 24, 2013

AEP-NRC-2013-77
10 CFR 50.73

Docket No.: 50-316

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
11555 Rockville Pike,
Rockville, MD 20852

Donald C. Cook Nuclear Plant Unit 2
LICENSEE EVENT REPORT 316/2013-001-00
Unit 2 Manual Reactor Trip

In accordance with 10 CFR 50.73, Indiana Michigan Power Company, the licensee for Donald C. Cook Nuclear Plant Unit 2, is submitting the following report as an enclosure to this letter:

LER 316/2013-001-00: "Unit 2 Manual Reactor Trip"

There are no commitments contained in this submittal.

Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P. Gebbie
Site Vice President

SJM/amp

Enclosure

c: J. T. King, MPSC
S. M. Krawec, AEP Ft. Wayne, w/o enclosure
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IE22
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NRC Form 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES 10/31/2013
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE

Unit 2 Manual Reactor Trip due to Lowering Steam Generator Level

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
07	28	2013	2013	- 001 -	00	09	24	2013	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
100	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Michael K. Scarpello, Regulatory Affairs Manager	(269) 466-2649

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	SB	TBG		N					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If Yes, complete 15. EXPECTED SUBMISSION DATE).* ☒ NO		09	26	2013

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On July 28, 2013, the Donald C. Cook Nuclear Plant Unit 2 reactor was operating at 100 percent power. At 1018, reactor operators manually tripped the reactor when reaching a low steam generator level threshold during a secondary plant transient event.

The secondary plant transient occurred when a steam supply valve closed to the Right Moisture Separator Reheater which resulted in feedwater heater level oscillations followed by Heater Drain Pumps tripping on heater low levels. This caused feedwater pump suction pressure to lower and automatically tripped the West Main Feedwater Pump. As a result, steam generator level lowered to 23 percent on the #4 Steam Generator. The reactor operators manually tripped the reactor based on a manual trip threshold for Steam Generator levels that was established by the Unit Supervisor.

The initiating cause of the steam supply valve closure and subsequent secondary plant transient was a loss of control air to the air operated valve resulting from fretting of the control air line.

The Reactor Protection System and the specified Auxiliary Feedwater System actuation was reported in accordance with 10 CFR 50.72(b)(2)(iv)(B) and 10 CFR 50.72(b)(3)(iv)(A). The valid actuation is reportable as a Licensee Event Report (LER) in accordance with 10 CFR 50.73(a)(2)(iv)(A).

NARRATIVE

Conditions Prior to Event

The Unit 2 reactor [RCT] was operating at 100 percent power.

Description of Event

On July 28, 2013, Cook Nuclear Plant (CNP) Unit 2 reactor was operating at 100 percent power.

At 0931, air operated valve 2-MRV-411, West Bypass Steam to Right Moisture Heating Coils Shutoff Valve MMO-431 Bypass Valve [SB][V], failed to the closed position from the normally full open position. Closure of the valve initiated a secondary plant transient, causing 2-HE-4A, Condensate Heater 4A [SD][HX], to reach a low-low level condition, which resulted in a trip of 2-PP-22N, North Heater Drain Pump [SN][P] at 0936. Operations reduced turbine load by 25 MW in response to the North Heater Drain Pump trip. At 1010 an Auxiliary Equipment Operator reported that valve 2-MRV-411 was locally observed in the closed position. Controller demand for 2-MRV-411 was observed full open and in manual control.

At 1012, heater level fluctuations occurring in 2-HE-4B, Condensate Heater 4B [SD][HX], resulted in a heater low-low level condition and tripped 2-PP-22S, South Heater Drain Pump [SN][P]. Operations initiated an additional turbine load reduction of 25 MW following the trip of South Heater Drain Pump. At 1017, 2-PP-1W, West Main Feedwater Pump [SJ][P], automatically tripped on low suction pressure.

All three Auxiliary Feedwater (AFW) pumps [BA][P] were started in accordance with the Loss of One Main Feedwater Pump procedure. The Unit Supervisor directed a manual reactor trip if steam generator levels lowered to 23% (automatic reactor trip setpoint is 22%). At 1018 Steam Generator #4 [SG] level lowered to 23% and Operators manually tripped the reactor.

All safety systems responded normally following the reactor trip with the exception of valve 2-FMO-221, Turbine Driven Auxiliary Feedwater Pump PP-4 Discharge to Steam Generator 2-OME-3-2 Control Valve [BA][V], which did not position as required, following the manual reactor trip. The affected discharge valve was normally open as expected during the AFW pump start and then closed instead of throttling to a preset intermediate position upon high flow rate. The Turbine Driven Auxiliary Feedwater Pump (TDAFP) [BA][P] was declared inoperable. Feedwater flow to 2-OME-3-2, Steam Generator #2 [SG], was maintained by 2-PP-3E, East Motor Driven Auxiliary Feedwater Pump [BA][P].

In accordance with 10 CFR 50.72(b)(2)(iv)(B), Event Notification 49220 was submitted on July 28, 2013, at 1343 to report the actuation of the Reactor Protection System. The specified system actuation of the AFW System was also reported in accordance with 10 CFR 50.72(b)(3)(iv)(A).

The manual reactor trip and the specified system actuation of the AFW System is reportable as a LER in accordance with 10 CFR 50.73(a)(2)(iv)(A).

NARRATIVE

Cause of Event

The initiating event was a failure of air operated valve 2-MRV-411 failing to the closed position due to fretting of the control air signal air line by a tubing support bracket.

The Root Cause of the event was determined to be a lack of procedural guidance and adequate controls for the Condensate Heater Condensate Bypass Control Valve, 2-CRV-224 [SD][V], automatic control functions.

Analysis of Event

The event was analyzed regarding the contributing factors leading up to the manual reactor trip on lowering steam generator levels caused by the low suction pressure automatic trip of the West Main Feedwater pump.

Investigation following the event concluded that 2-CRV-224 [SD][V] did not open to help prevent a low feedwater suction pressure from tripping the main feedwater pump. The associated controller [SD][PMC] was found at a setpoint lower than the intended design, which prevented 2-CRV-224 from opening. Controller upgrades occurred in 1995 and 2003 that changed the methodology for setpoint inputs. The root cause was stated as a lack of procedural guidance and controls for the valve controller functions following controller upgrades.

A failure of 2-LPD-320N, North Heater Drain Pump PP-22N Discharge Check Valve [SN][CKV], was discovered following the manual reactor trip. The check valve was found in a partially open position which allowed a diversion of a portion of condensate flow away from the main feedwater pump suction.

The TDAFP discharge valve that did not automatically position as expected following the manual reactor trip was found with limit switches adjusted incorrectly. The incorrect limit switch settings did not affect the main feedwater pump suction during the event or impact feedwater flow to #3 steam generator. The limit switches were adjusted and tested satisfactory.

A risk impact review for this event indicates the trip was uncomplicated. All control rods [JC][JD] inserted, the main turbine-generator [TB][TG] tripped, offsite power transferred and remained energized by the station Reserve Auxiliary Transformers [EB][XFMR] as designed. No safety injection or engineered safety feature actuations occurred or were warranted beyond those expected for a nominal reactor trip. Main feedwater function was recoverable if it had been needed. The control room operators did not enter any additional emergency operating procedures after the trip, except those optimally expected. The operators noted that 2-FMO-221 (TDAFP Discharge to #3 SG flow control valve) ran fully closed rather than stopping at its intermediate flow retention setting. Discussion with the control room operators afterward and review of the Reactor Trip Report indicates there was sufficient flow from the motor driven AFW pumps such that the closed AFW valve had no significant impact on plant trip response. The operators were not required to manually reopen valve 2-FMO-221 immediately following the trip. The operators subsequently opened 2-FMO-221 to re-align the TDAFP to an available status and the valve did respond as expected to manual operation. The operators could have opened the valve during the event for additional AFW flow to #3 SG if needed.

Overall, the trip was an uncomplicated event with a malfunction of one AFW valve. The crew had observed and was aware of the AFW valve condition, and could have manually used the valve if needed. For these reasons, this trip did not pose any significant risk.

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NARRATIVE

Corrective Actions

Completed Corrective Actions

Plant procedures were revised to provide guidance to establish and maintain condensate bypass valve 2-CRV-224 controller setpoint to modulate beginning at 240 psig.

North Heater Drain Pump discharge check valve 2-LPD-320N was replaced.

Failed control air tubing supply line to 2-MRV-411 was replaced.

The motor operated valve actuator limit switch settings for 2-FMO-221 were adjusted and tested.

Planned Corrective Actions

No additional corrective actions are planned.

Previous Similar Events

LERs for CNP Unit 1 and Unit 2 were reviewed for the previous five years and found no similar events.